

Reactor Pressure Vessel Embrittlement Monitoring and Prediction in Long-Term Operation

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Reactor Pressure Vessel Embrittlement Monitoring and Prediction in Long-Term Operation

Public Meeting
October 18, 2021

Agenda

- Welcome / Introductions / Logistics
- NRC Presentation: Reactor Pressure Vessel Embrittlement Monitoring and Prediction in Long-Term Operation
- Public Presentations
- Discussion and Q&A
- Closing Remarks, and Adjourn

Welcome & Introductions

Welcome

- Robert Taylor, NRR Deputy Office Director

Introductions

- David Rudland, NRR Senior Technical Lead
- Allen Hiser, NRR Senior Technical Lead
- Stewart Schneider, NMSS Senior Project Manager
- Joan Olmstead, NRC Facilitator



Meeting Logistics



- Meeting visuals are through WebEx.
- Meeting audio is through the bridgeline.
- Participants are in listen-only mode until the question and answer period. The operator will open phone lines during this time.
- This is an Information Meeting with a Question and Answer Session. The purpose of this meeting is for the NRC staff to meet directly with individuals to discuss regulatory and technical issues. Attendees will have an opportunity to ask questions of the NRC staff or give feedback about the issues discussed after all presentations; however, the NRC is not actively soliciting comments towards regulatory decisions at this meeting.
- This meeting is being transcribed. The transcript will be available through the meeting summary.

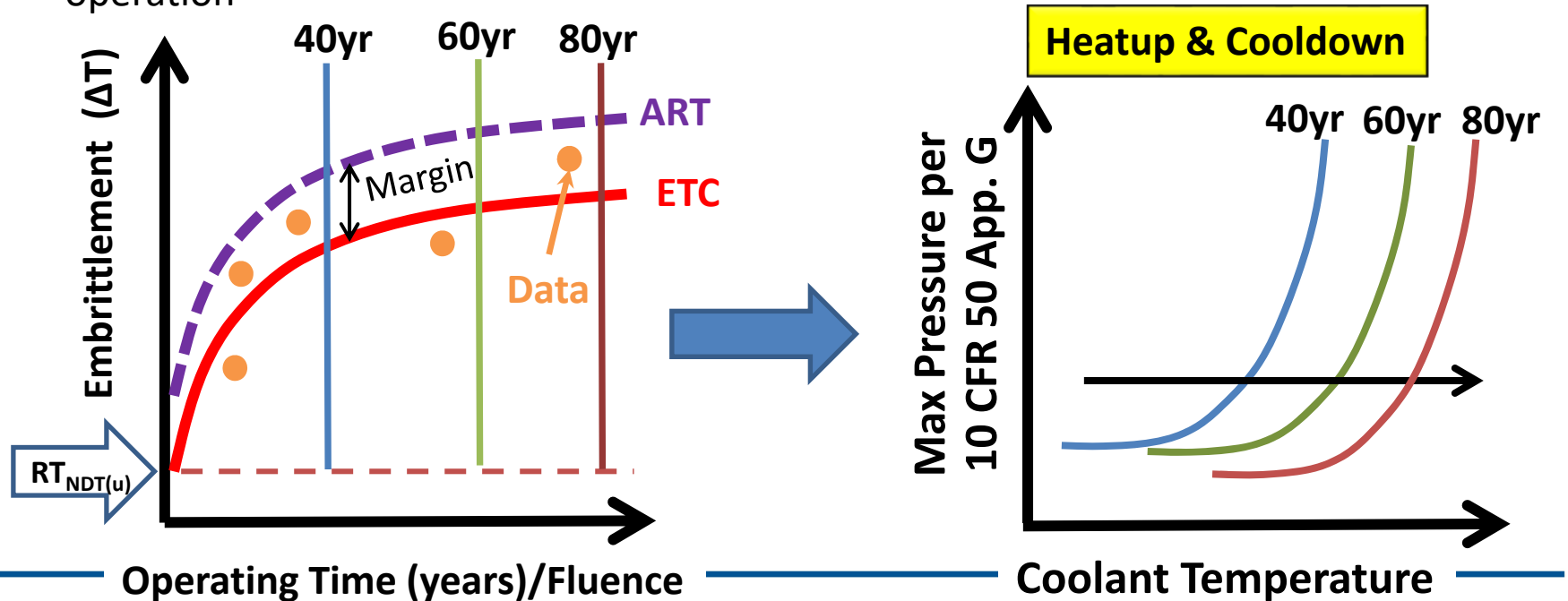
Meeting Purpose

- Continue discussion of issues from May 2020 public meeting (ML20168A008)
 - Regulatory Guide 1.99 Rev 2 (RG 1.99) and 10 CFR 50.61 embrittlement trend curve
 - Appendix H surveillance testing
- Discuss a holistic risk-informed analysis of these issues and its potential impact on reactor pressure vessel (RPV) integrity
- This is a technical discussion; no regulatory decisions will be made at today's meeting
- NRC staff would like feedback on analysis approach and results

Background

Monitoring and Prediction of Embrittlement

- Embrittlement Trend Curve (ETC) provides estimates of change in fracture toughness (ΔT or ΔRT_{NDT}) as a function of fluence
- Surveillance capsule testing provides monitoring to ensure ETC predicts plant specific behavior properly
- Together they are used to determine pressure-temperature (PT) limits for normal operation



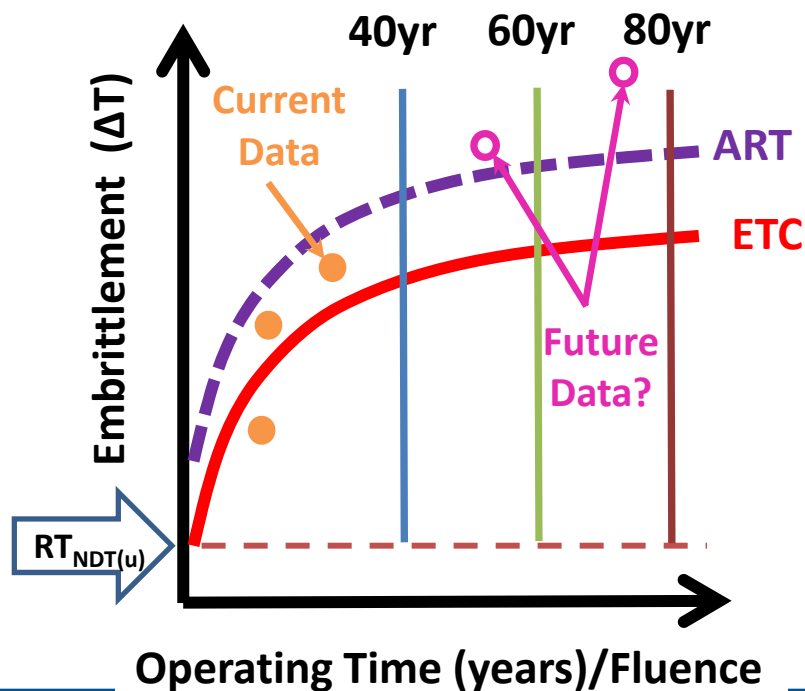
ART = Adjusted Reference Temperature

Ideal Scenario

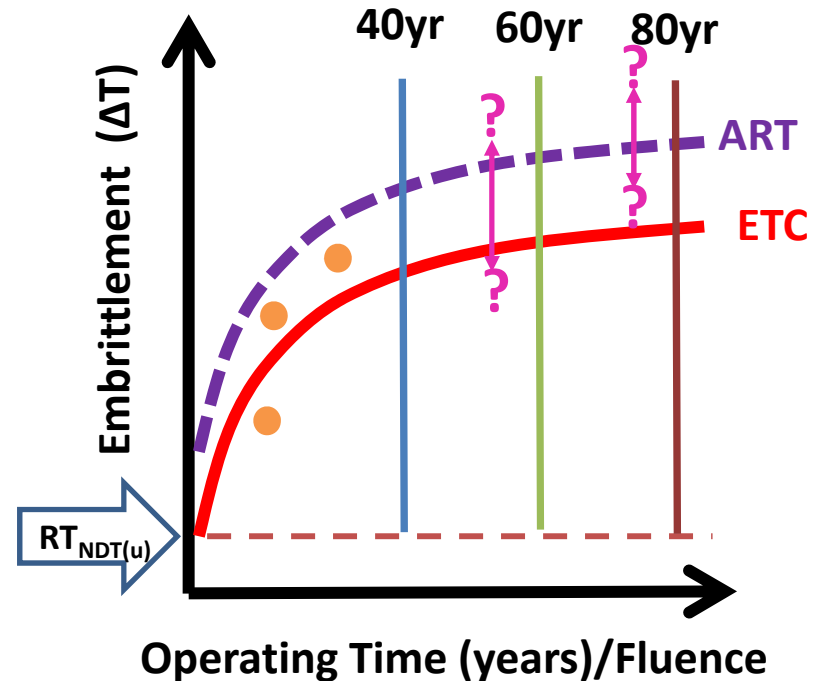
- ETC provides conservative predictions of embrittlement
- Surveillance data covers all operating periods

Potential Uncertainty Sources

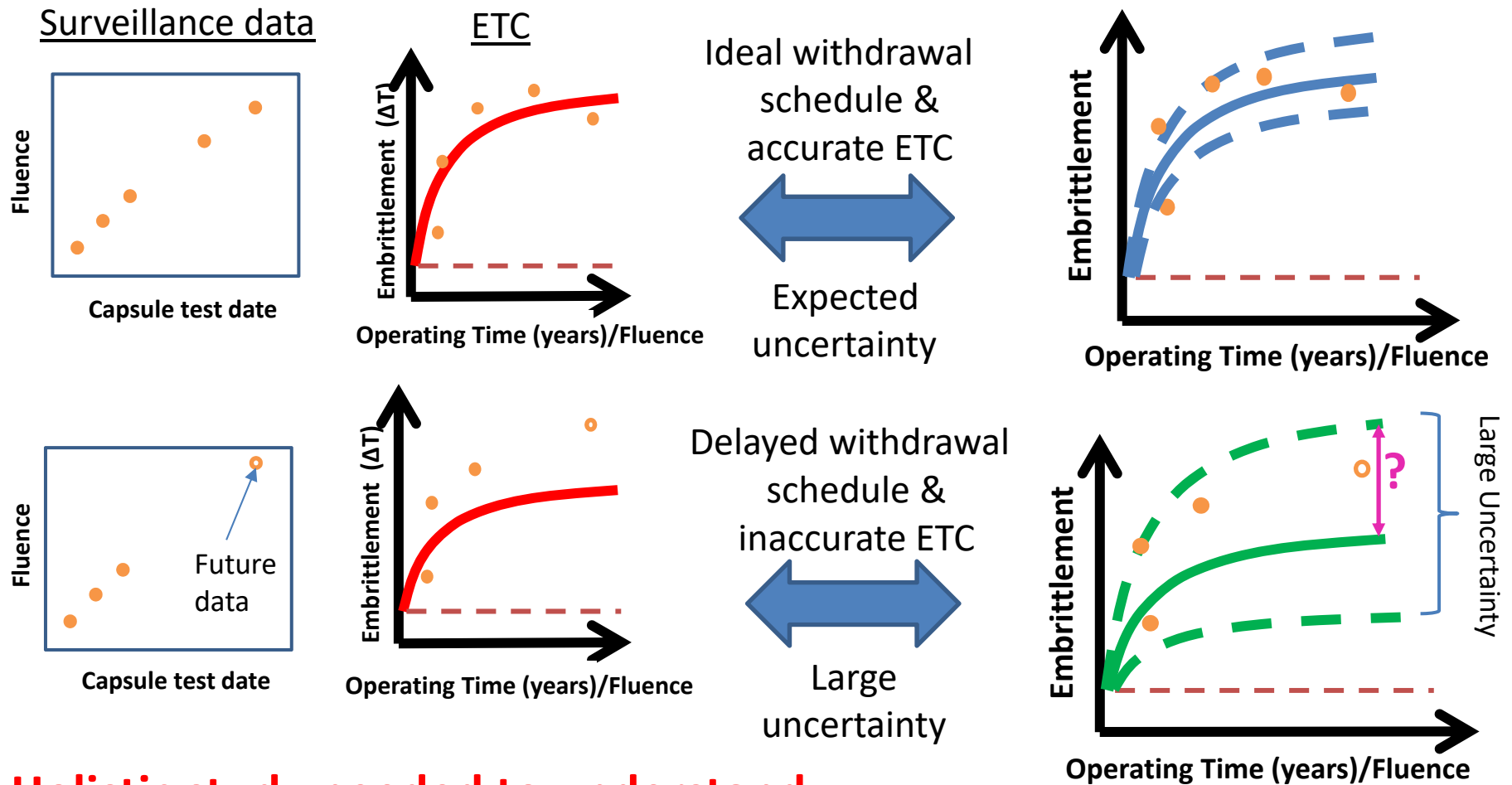
IF ETC under-predicts measurements



IF Limited Surveillance Data is Available



Embrittlement Uncertainty



Holistic study needed to understand impact of uncertainty

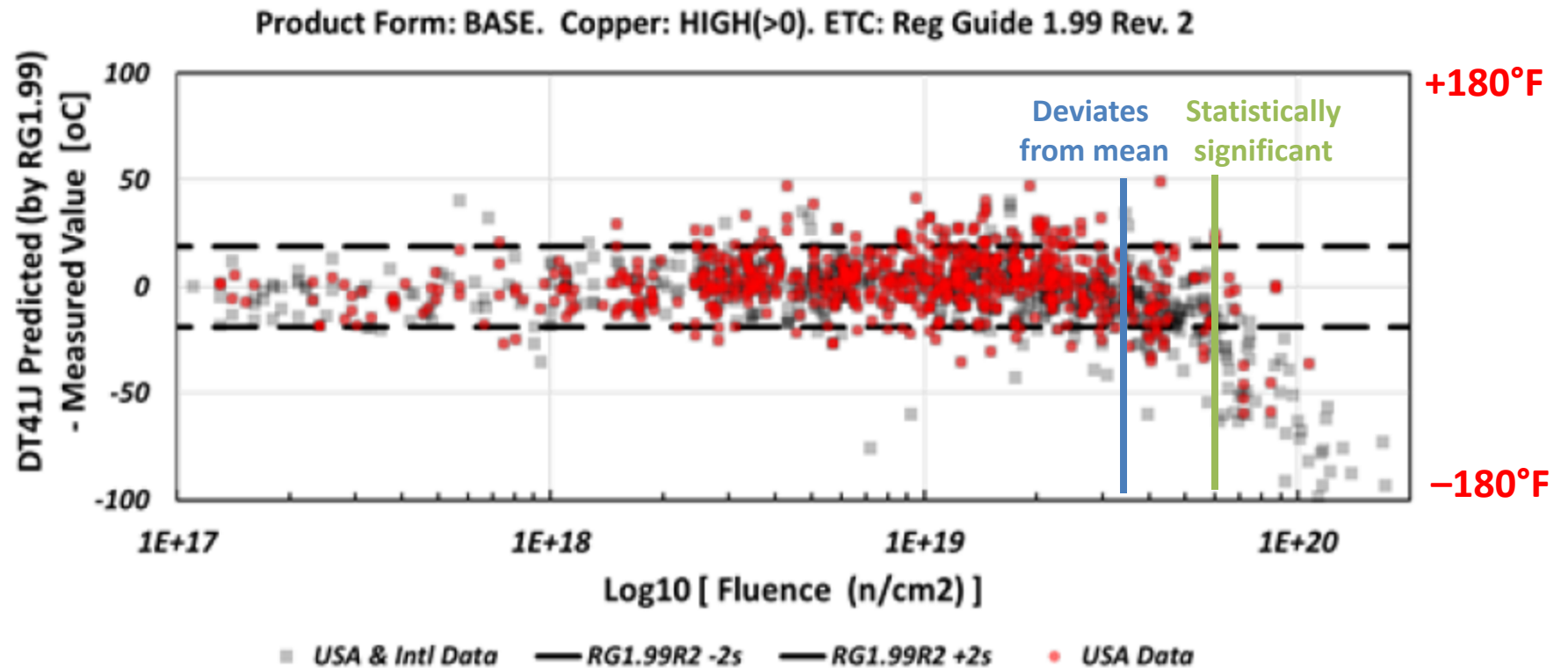
Current Perspective of Potential Issue

- High confidence that currently operating plants remain safe
 - Recent licensing actions remain valid
 - Insufficient embrittlement monitoring and under predictions of reactor vessel embrittlement will eventually (after about 10 years) impact the staff's confidence in the integrity of the reactor pressure vessel in long-term operation, i.e., both safety margins and performance monitoring may be impacted
 - Further work is needed to determine which plants are impacted by this potential issue
-

Embrittlement Trend Curve

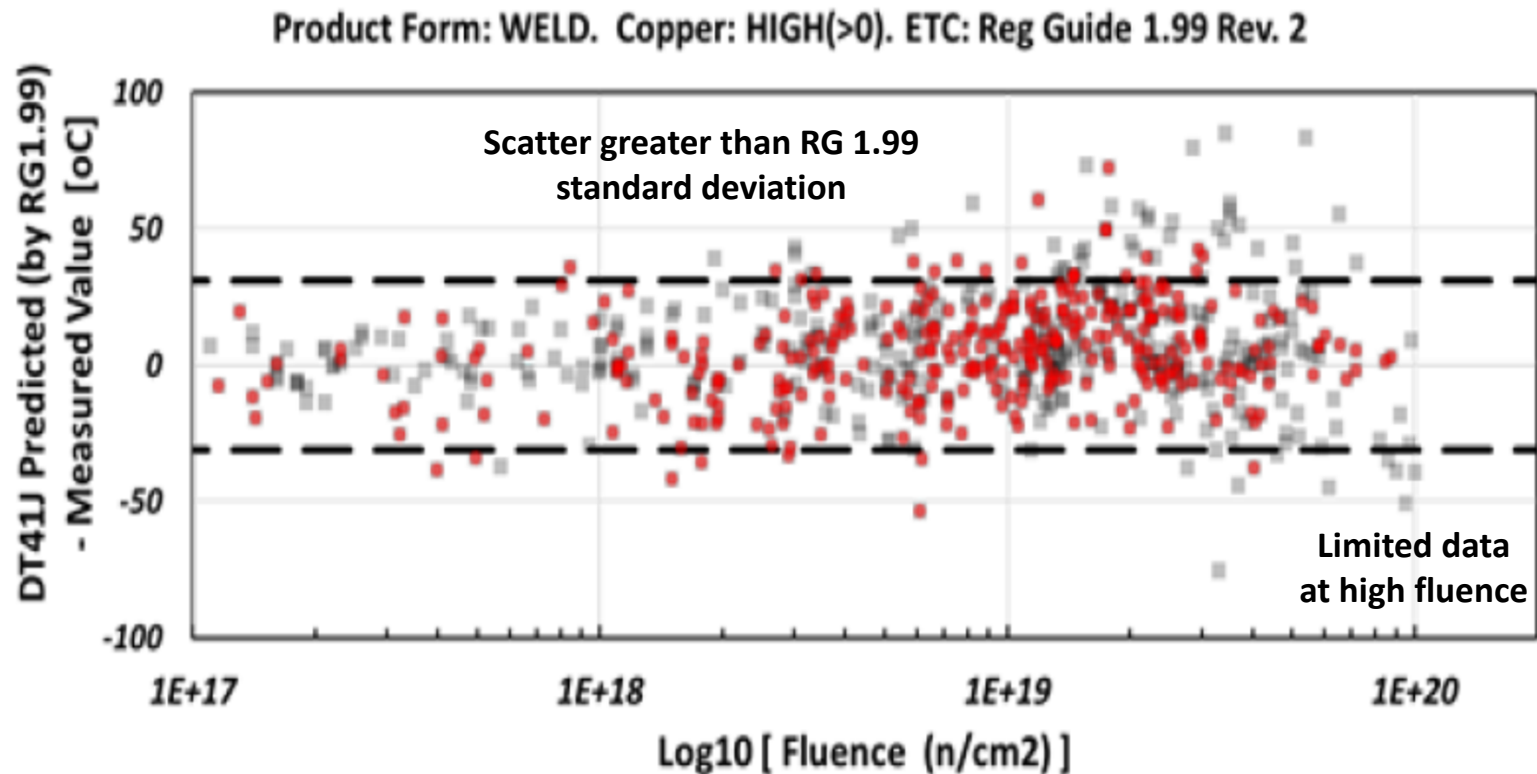
- May 1988, NRC published RG 1.99, which contained an improved embrittlement trend curve (ETC)
 - Fit based on 177 datapoints
- June 1991, NRC updated 10 CFR 50.61 to include the ETC from RG 1.99
 - Addressed lower than measured predictions (up to 60°F) of embrittlement in some vessels
- This ETC was re-evaluated for continued adequacy in 2014 (ML13346A003) and in more detail in 2019 (ML19203A089)

Issue – ETC

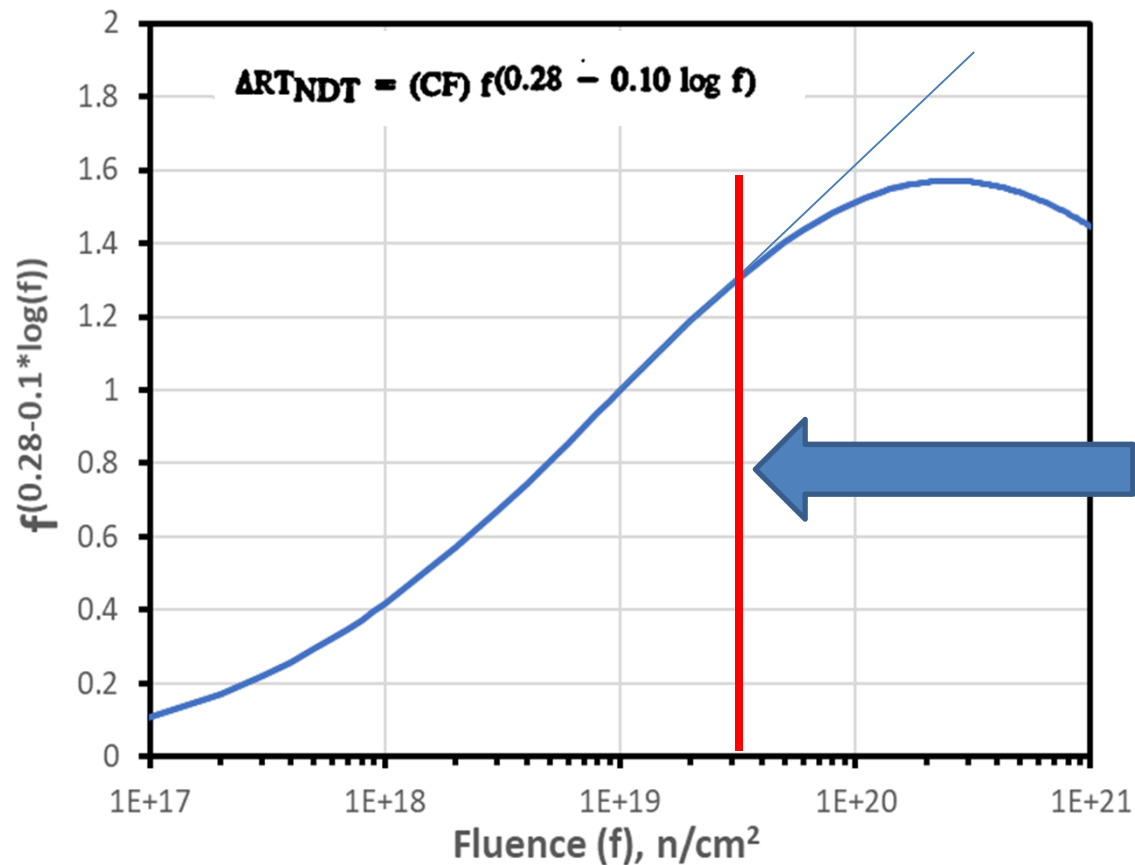


DT41J = ΔT_{41J} is a measurement of embrittlement representing the shift in transition temperature from brittle to ductile fracture at an impact toughness of 41J

Issue – ETC



Issue – ETC Fluence Function



Fluence function begins to “flatten” at the same fluence level underprediction occurs in Slide 12

Surveillance Capsule Delays

- Appendix H to 10 CFR Part 50 requires periodic monitoring of changes in fracture toughness caused by neutron embrittlement
 - ASTM standard (E185-82) allows final capsule fluence to be 2X RPV “design” fluence – plants change (intended 40-year) design fluence to current license length (e.g., 60 or 80 years)
 - ASTM standard (for 40 years) permits holding last capsule without testing
- Commission finding (“Perry decision” NRC Administrative Letter 97-04) that staff review of requests to change capsule withdrawal schedules is limited to verification of conformance with the ASTM standard (i.e., not based on technical or safety considerations)
 - Capsule withdraw and testing repeatedly delayed in some cases to achieve higher fluence

License Renewal

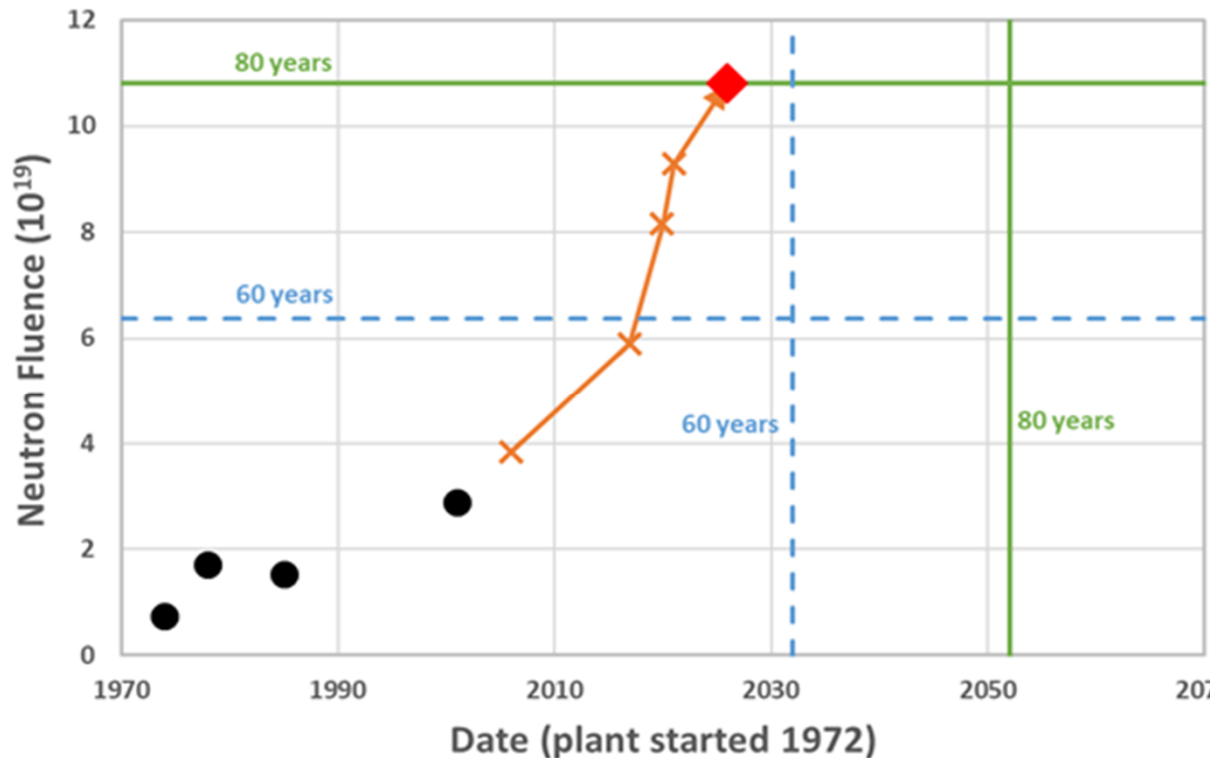
- Regulations are unchanged; surveillance program addressed in guidance
 - Guidance provides flexibility for licensees to demonstrate adequate management of RPV embrittlement due to varying plant-specific circumstances
- Aging Management Program XI.M31, “Reactor Vessel Material Surveillance”
 - Continues reliance on Appendix H program using ASTM E185-82
 - GALL Report (NUREG-1801, Rev. 1) for license renewal (40 to 60 years)
 - “shall have at least one capsule with a projected neutron fluence equal to or exceeding the 60-year peak reactor vessel wall neutron fluence prior to the end of the period of extended operation”
 - Describes use of reconstituted specimens and use of operating restrictions (neutron flux, spectrum, irradiation temperature, etc.)
 - GALL-SLR Report (NUREG-2191) for subsequent license renewal (60 to 80 years)
 - “withdrawal and testing of at least one capsule . . . with a neutron fluence of the capsule between one and two times the peak neutron fluence of interest at the end of the subsequent period of extended operation” – or data from a prior tested capsule
 - Specifies – “it is not acceptable to redirect or postpone the withdrawal and testing of that capsule to achieve a higher neutron fluence that meets the neutron fluence criterion for the subsequent period of extended operation”

License Renewal in Practice

- Licensees have changed capsule withdrawal schedules prior to application for license renewal or subsequent license renewal
 - Change is evaluated under current approach of “conformance verification”
- (Updated) current licensing basis surveillance program for license renewal/subsequent license renewal is then consistent with the program in GALL/GALL-SLR

Issue – Appendix H

Performance Monitoring



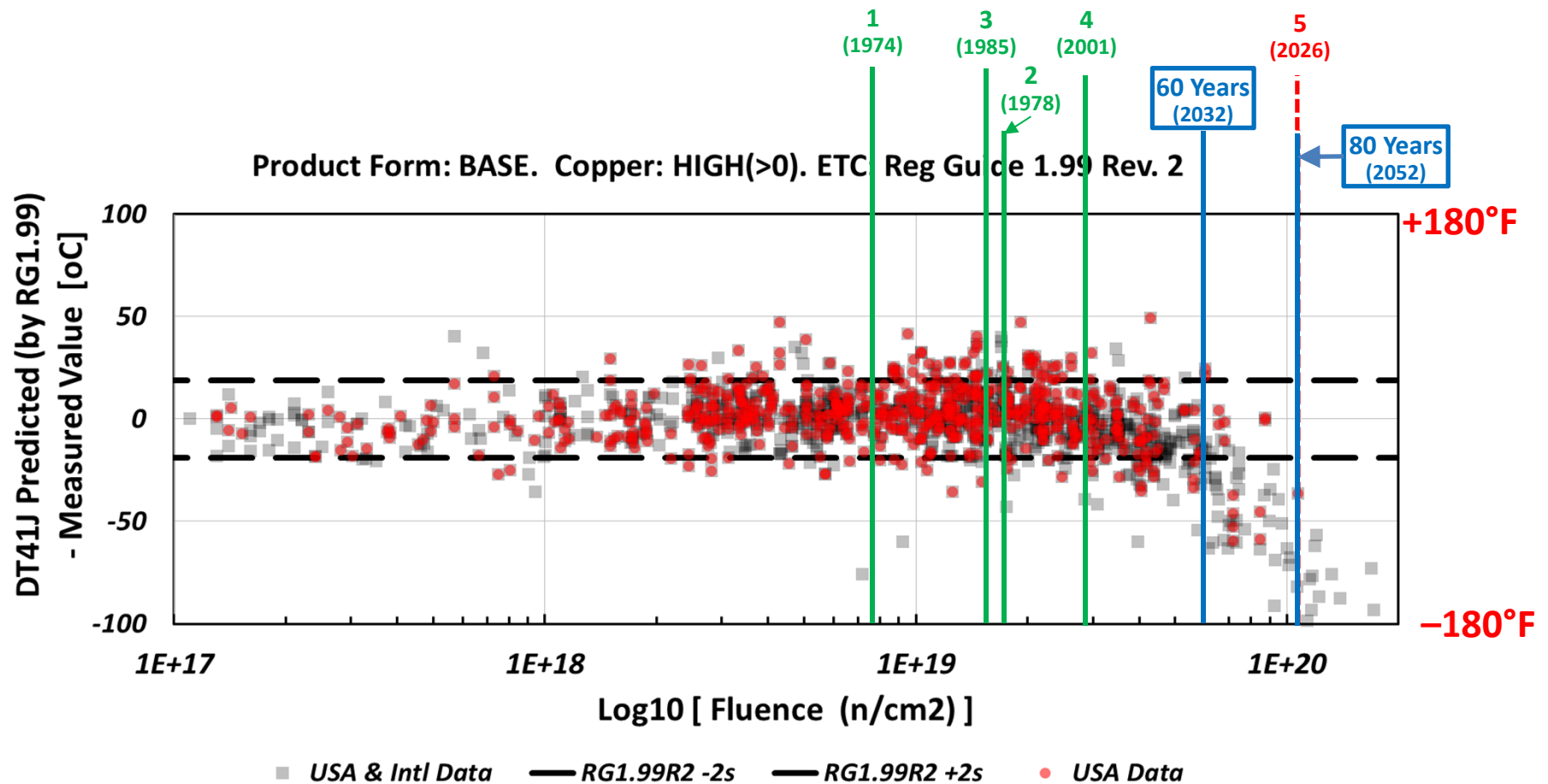
Many licensees have delayed capsules (time and/or fluence), some recent examples:

Plant	Capsule #	# of times delayed
Turkey Point	5	4
Robinson	5	2
Surry U1	5	2
Surry U2	5	2
North Anna U1	4	2
North Anna U2	4	2
St. Lucie U2	4	1
Point Beach	5	1

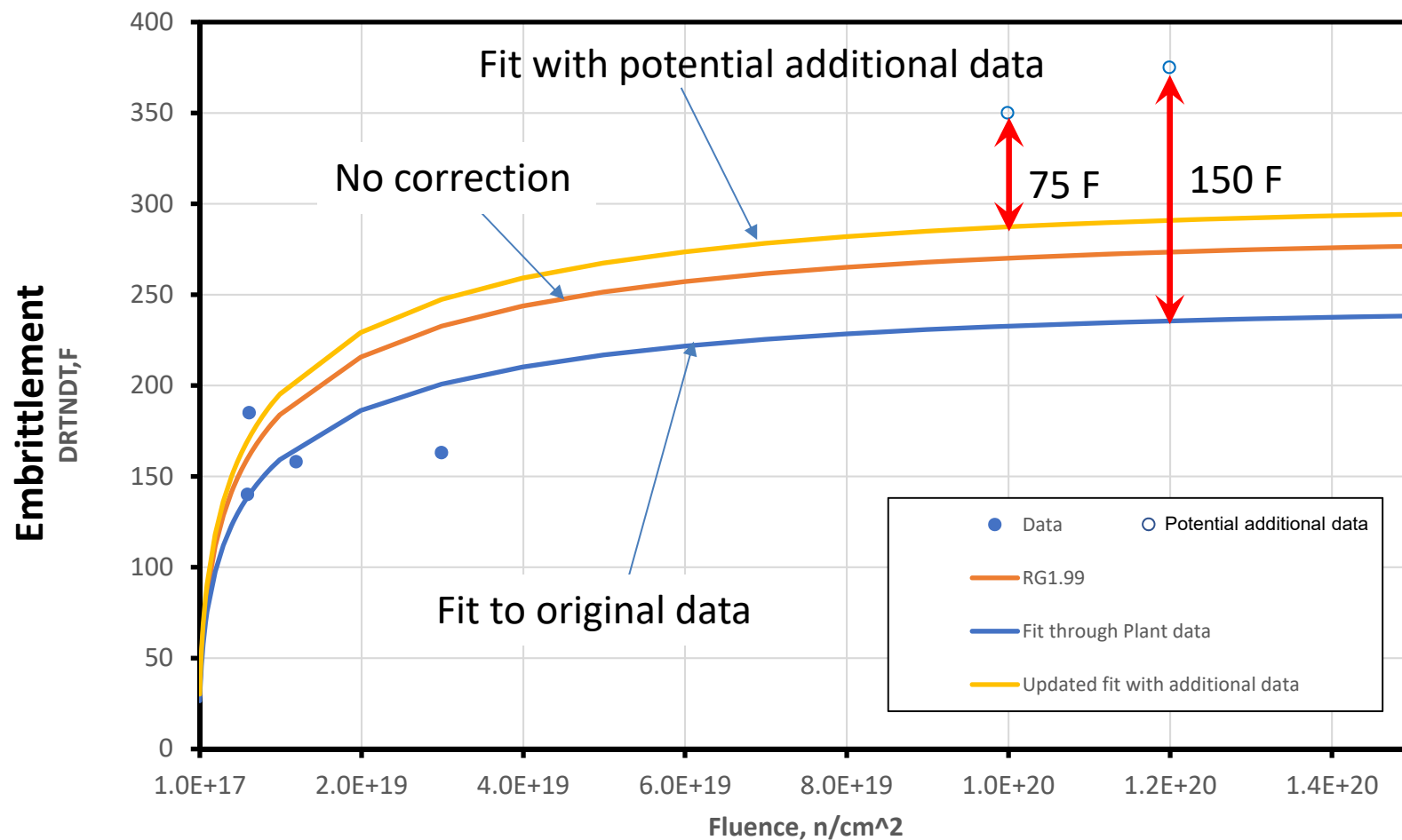
Capsule withdrawal schedule changes include delays in both time and/or fluence

Not all plants have delayed withdrawal of capsules

Potential Impact of Issue



Potential Impact of Issue



Risk-informed Analysis

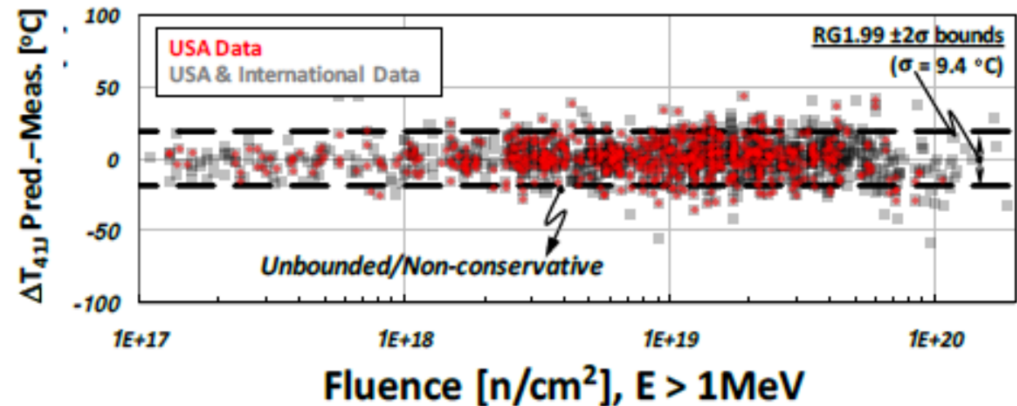
- Considered combined effects of surveillance and embrittlement predictions
- Leveraged 5 principles of risk-informed decision making
- Targeted sample of plant data used, but much plant specific information not available



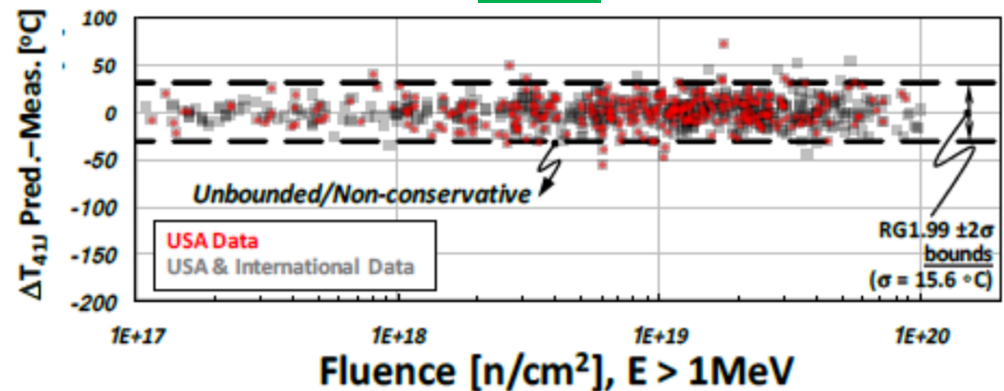
Analysis Assumptions

- Comparisons based on ASTM E900-15 ETC
- The NRC staff found that the ASTM E900-15 ETC provided the most accurate characterization of this database*

Plates and Forgings



Welds



*"Basis for a Potential Alternative to Revision 2 of Regulatory Guide 1.99,"
TLR-RES/DE/CIB-2020-11, ML20345A003

Analysis Assumptions – Fleet Impact Study

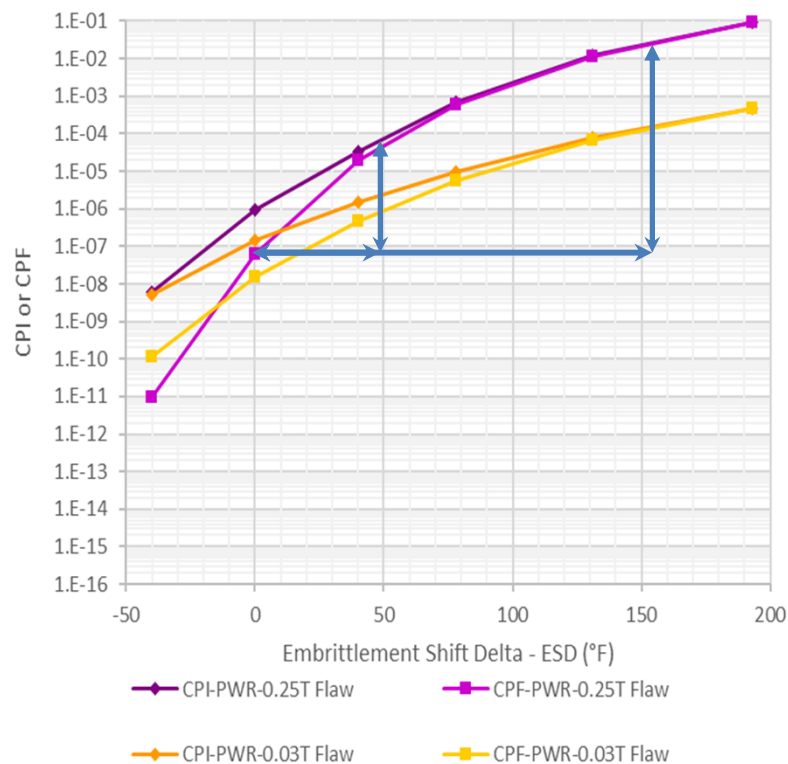
- A targeted sample of 21 plants
- Emphasis on high fluence plants, with a few low Cu plants and BWRs to round out
- Determined changes in adjusted reference temperature resulting from switching ETCs – “embrittlement shift delta” (ESD)
- Results used to benchmark ESD range of risk analysis

Results – Fleet Impact Study

- There is a tendency for material reference temperatures to increase when switching from RG 1.99 to ASTM E900-15.
- Base materials are more likely to see increases in reference temperatures than weld materials.
- Only a handful of plant limiting materials will have ESDs > 50 °F, and these tend to be at fluences $\sim 6 \times 10^{19}$ n/cm².
- Range of ESDs assumed in risk study bounds fleet impact findings.

Risk of Failure

PT100-100 Cooldown
ARTmax(PWR-no shift) = 260.7°F



ESD represents the underprediction of ΔRT_{NDT}

Large Uncertainties:

- Unknown frequency of transient
- Actual plant fluence variations
- Are these analyses bounding?
 - Unknown plant-specific considerations
- How much protection do administrative and other operational limits provide against violating the PT limit?

“RG 1.99 Revision 2 Update FAVOR Scoping Study,”
 May 6, 2021, TLR RES/DE/CIB-2020-09, Rev. 1,
 ML21126A326

Through-Wall Crack Frequency Results

Transient Type	Shallow Flaw	1/4T Flaw	Comment
BWR P-T Limit Cooldowns	$CPF \leq 1 \times 10^{-6}$ for all ESDs	$CPF \geq 1 \times 10^{-6}$ for ESD > 40 °F	BWRs must cooldown on saturation curve, so cooldown on licensed limits not plausible.
BWR Saturation Cooldown	$CPF \leq 1 \times 10^{-6}$ for all ESDs	$CPF \leq 1 \times 10^{-6}$ for all ESDs	
BWR Leak Test, Cooldown rate ≤ 50 °F/hour	$CPF \leq 1 \times 10^{-6}$ for all ESDs	$CPF \geq 1 \times 10^{-6}$ for ESD > 100 °F	Additional information is desired to determine if high cooldown rates are possible, or ASME Code action will be pursued to prohibit.
BWR Leak Test, Cooldown rate > 50 °F/hour	$CPF \leq 1 \times 10^{-6}$ for all ESDs	$CPF \geq 1 \times 10^{-6}$ for ESD > 100 °F	
PWR P-T Limit Cooldowns	$CPF > 1 \times 10^{-6}$ for ESDs ≥ 50 °F	$CPF > 1 \times 10^{-6}$ for ESD ≥ 20 °F	Additional information on event frequencies is desired to confirm $TWCF < 1 \times 10^{-6}$ /year.
PWR Cooldown, Actual Transients	$CPF < 1 \times 10^{-6}$ for most transients	n/a	

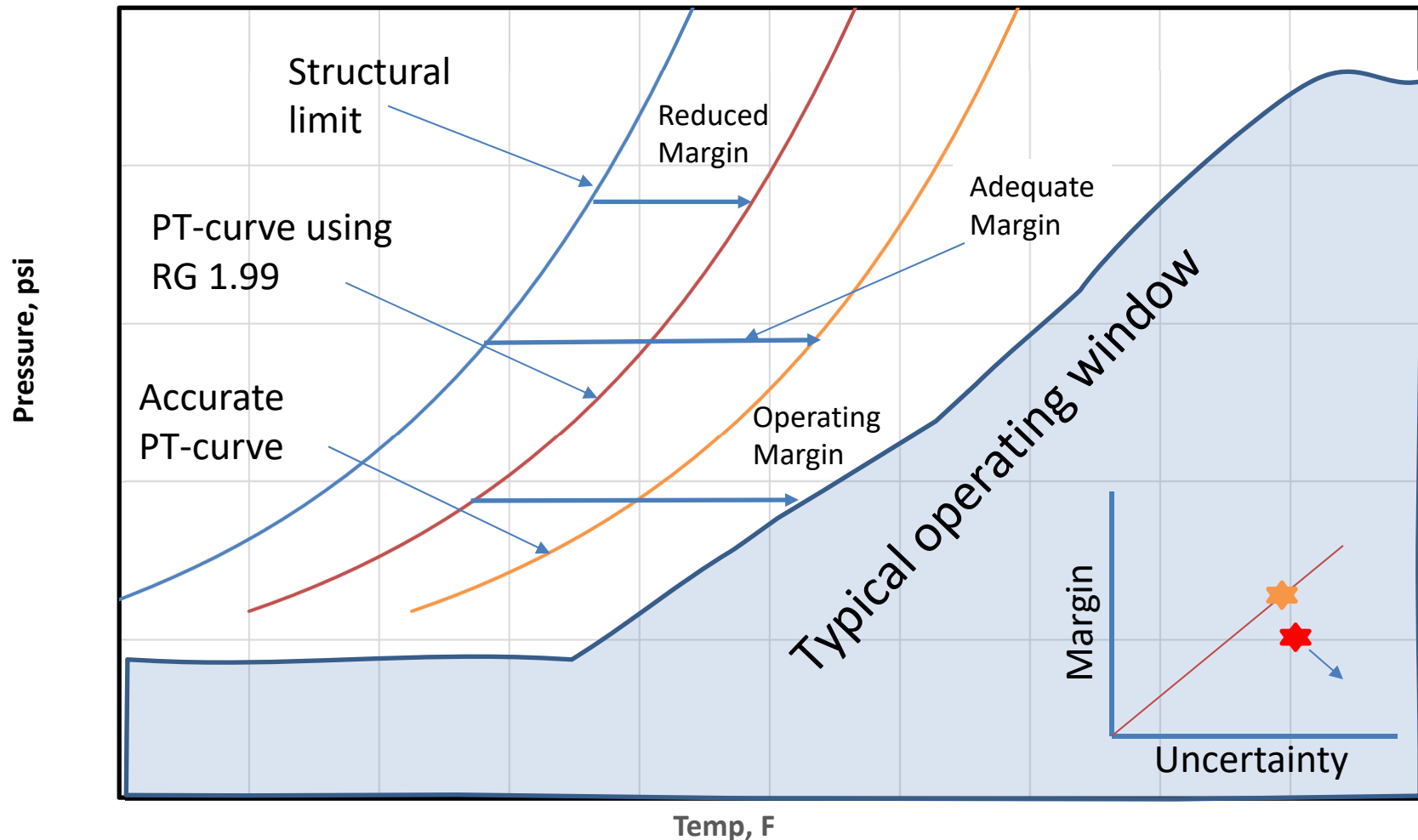
Pressurized Thermal Shock Considerations

- 10 CFR 50.61 uses ETC from RG 1.99
- RT_{PTS} from 10 CFR 50.61 might be impacted
 - Limits of 270 °F for plates, forgings, and axial weld materials, and 300 °F for circumferential weld materials
- However, through-wall crack frequency calculated with corrected embrittlement less than 1×10^{-6} for all cases investigated

Safety Margins

- Uncertainties in risk calculations are high and increasing with time
- Even though the risk appears low, resolving these issues will help maintain the fundamental safety principles that are the basis of plant design and operation
- Safety margins, as provided by regulations and current license bases, provide reasonable assurance against brittle fracture

Safety Margins Illustration



Uncertainties increasing due to lack of surveillance, but margin is less due to embrittlement underprediction

Performance Monitoring

- Performance monitoring ensures
 - Analysis results remain valid with time
 - No unexpected (or unmodelled) adverse safety issue occurs
- Delaying capsule withdrawal for an extended period with the possibility of no future data represents a lack of performance monitoring

Analysis Summary

- With the current state of knowledge, a generalized analysis suggests the overall risk of brittle fracture is low
- The uncertainty in these results is high and increases with time
 - Plant specific details not considered
- Under certain conditions, safety margins are impacted and are decreasing as uncertainty increases
- Delaying capsules at high fluence represents a lack of sufficient performance monitoring
- Issues are plants with fluences $> 6 \times 10^{19} \text{ n/cm}^2$

Who is Impacted?

- Embrittlement Underprediction

Percentage of Fleet Surpassing Fluence Levels			Percentage of PWRs Surpassing Fluence Levels	
Year\Fluence	$6 \times 10^{19} \text{ n/cm}^2$	$8 \times 10^{19} \text{ n/cm}^2$	$6 \times 10^{19} \text{ n/cm}^2$	$8 \times 10^{19} \text{ n/cm}^2$
60 years	6%	0%	9%	0%
80 years	22%	10%	34%	15%

- Plant specific details (e.g., limiting material, etc.) may contribute to which plants are impacted
- More work is needed to determine which plants are impacted

- Lack of Surveillance Data

- Any plant renewing license that chooses to delay last capsule

Staff Goals

- Currently, regulations are sufficient for reasonable assurance of adequate protection against brittle fracture of vessel
- Staff wants to ensure continued reasonable assurance in long-term operation
 - Provide remedies for the identified issues with RPV surveillance requirements and embrittlement predictions, on a risk-informed, performance basis
- Do not impact those plants that are not adversely affected by the issues
 - Plant-specific surveillance data that covers end of license fluence level
 - Projected fluence at end of license $< \sim 3 \times 10^{19} \text{ n/cm}^2$

Options to Meet Goal

- Plant-specific action
- Focused regulatory action
- Generic communication
- No action

Discussion Topics

- Is the staff's approach to determine the safety impact of the surveillance and embrittlement issues appropriate?
- What other options could be considered to address these issues?
- Are there other potential adverse impacts to plant operations (e.g., unnecessary updates to PT limits) that should be considered?
- Is now the right time to pursue these issues?

Reminder: The NRC is not actively soliciting comments towards regulatory decisions at this meeting.

Summary



- High confidence that currently operating plants remain safe, and recent licensing actions remain valid
- Issue will eventually (after about 10 years) impact the staff confidence in the integrity of the reactor pressure vessel in long-term operation, i.e., both safety margins and performance monitoring may be impacted
- Further work is needed to determine which plants are impacted by this issue
- Proactively ensure continued reasonable assurance through a risk-informed, performance-based solution
 - Staff is considering options – desires focused solution to only those conditions adversely impacted by this issue


Public Presentations

Discussion




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- NRC Public Meeting Feedback Form
 - Link to the form is available at the public meeting website under the meeting notice
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Thank You